NON-PUBLIC?: N
ACCESSION #: 8806290030
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Sequoyah, Unit 2 PAGE: 1 of 5

DOCKET NUMBER: 05000328

TITLE: Reactor Trip Resulting From Low Reactor Coolant System Flow Signal Caused By A Procedure Noncompliance
EVENT DATE: 05/23/88 LER #: 88-024-00 REPORT DATE: 06/17/88

OPERATING MODE: 1 POWER LEVEL: 000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
N
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SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT: On May 23, 1988, at 0028 EDT with unit 2 at 70 percent power, a reactor trip occurred from a low flow signal on reactor coolant system (RCS) loop 4 (2 out of 3 channels tripped on any one RCS loop above 35 percent power). At the time of the trip, Surveillance Instruction (SI)-246, "Recalibration Procedure for Reactor Coolant Flow Channels," was in progress to recalibrate the loop 4, channel II transmitter (2-FT-68-71B). The reactor trip "Sequence of Events Record" showed the trip being initiated from RCS loop 4, channel III bistable 2-FS-68-71D. Investigation into the cause of the trip revealed that the transmitter being calibrated was attached to a common sense line with loop 4, channel III transmitter, 2-FT-68-71D. It was discovered that the instrument mechanics (IMs) performing SI-246 had not complied with procedure when valving out the transmitter 2-FT-68-71B. SI-246 instructs the performer to relieve RCS system pressure by cracking open the transmitter high side test tee. Contrary to this instruction the IMs relieved system pressure through the drain valve which routes to a closed drain system which made it impossible to determine the amount of fill fluid (RCS water) lost when the drain valve was open. It was theorized that when the final step was performed to open the high side isolation valve, the void in the drain line caused a pressure drop in the common sense line and subsequent drop in the output of 2-FT-68-71D. This drop in the 2-FT-68-71D output caused the low flow bistable 2-FS-68-71D to actuate and since the 2-FS-68-71B bistable was already tripped under SI-246, this completed the necessary 2 out of 3 logic for the reactor

trip. A reenactment of the procedure steps, confirmed the theory as discussed above and proved the reactor trip was caused by the procedure deviation. To provide assurance that an adverse trend does not exist on common interactions of equipment, TVA will review past reactor trips at SQN for a similar occurrence of a common interaction between equipment that has caused a reactor trip.

(End of Abstract)

DESCRIPTION OF EVENT

On May 23, 1988, at approximately 0028 EDT with unit 2 in mode 1 (70 percent power, 2240 psig, 567 degrees F), a reactor trip occurred on unit 2. The trip was a result of a reactor coolant system (RCS) (EIIS Code AB) low flow signal in RCS loop 4 (two out of three channels tripped on any one RCS loop when above 35 percent power). Before the trip, performance of Surveillance Instruction (SI)-246, "Recalibration Procedure for Reactor Coolant Flow Channels," was in progress to recalibrate nine of the 12 RCS flow transmitters (3 transmitters per RCS loop). This recalibration was a result of RCS flow values calculated in SI-155, "Reactor Coolant Flow Verification," differing from the existing calibration values on 9 of the 12 RCS flow transmitters (2-FT-68-6A, -6B, -6D, -29A, -29B, -48D, -71A, -71B, and -71D).

SI-155 is performed every 18 months in accordance with Technical Specification (TS) Surveillance Requirement (SR) 4.2.3.5 to measure the total RCS flow rate. The RCS full (100 percent) flow rate is measured by means of heat balance (calorimetric) calculations performed on the steam generators. The calculated 100 percent RCS flow values in SI-155 are then compared with existing transmitter calibration values in SI-246. If the transmitter calibration values are within plus or minus three percent of the calculated values derived in SI-155, then no recalibration is performed; however, any transmitter values not within this acceptable range are recalibrated under SI-246. As mentioned above, the latest performance of SI-155 on May 13, 1988, had revealed that the existing calibration values on 9 of the 12 RCS flow transmitters were outside the acceptable range when compared with the calculated values.

Before the reactor trip, four of the nine transmitters (2-FT-68-6A, -6B, -29A, and -71A) outside the acceptable range had been successfully recalibrated between May 21 and May 23, 1988, with no adverse effects. The recalibration of 2-FT-68-71B (loop 4, channel II) was near completion when the reactor trip occurred. The transmitter had been removed from service at 2216 EDT on May 22, 1988, and the channel bistable 2-FS-68-71B (RCS low flow) was placed in the tripped condition at 2226 EDT. The transmitter had already been valved out of service, recalibrated, and was being valved back in service when the trip occurred. A review of the reactor trip, "Sequence of Events Record," printout showed that the RCS loop 4 low flow bistable (2-FS-68-71D) had actuated to start the reactor trip sequence of events.

This actuation was noted by computer log point F0462D (RCS, loop 4, channel III, Low Flow Part Reactor Trip) showing a printout. This log point is activated from the output of loop 4, channel III bistable 2-FS-68-71D. The actuation of this bistable concurrent with the previously tripped bistable 2-FS-68-71B completed the necessary reactor trip logic (EIIS Code JC) of two out of three low RCS flow from any one loop coincident with permissive P-8 (above 35 percent power). The reactor first out annunciator, "One Loop Low Flow - Reactor Trip,"

was received in the main control room, but the bistable status light for 2-FS-68-71D was not noted at the time of the trip. A subsequent review of the "Sequence of Events Record" printout showed that the bistable 2-FS-68-71D was only in the tripped condition for approximately .068 seconds, which would not have allowed the status light to be energized for a sufficient duration to be noticed.

An immediate investigation ensued to determine the cause of the bistable 2-FS-68-71D actuation. After reviewing drawings of the transmitter sense line configuration, it was noted that all three RCS flow transmitters on loop 4 (FT-68-71A, -71B, and -71D) share a common sense line on the transmitter high side. The SI-246 procedure was reviewed, and the instrument mechanics (IMs) performing the SI were interviewed to determine if an interaction may have occurred in the common sense line while calibrating 2-FT-68-71B, that may have affected the 2-FT-68-71D transmitter. After reviewing SI-246 and interviewing the IMs, it was discovered that the SI-246 procedure had not been complied with in the valving out of the transmitter 2-FT-68-71B. The procedure is written to calibrate the transmitter while "on line" without losing any of the RCS fill fluid in the sense line (i.e., wet calibration). Appendix A-11, step 5.2.3, of SI-246 instructs the performer to "crack the transmitter high side test tee fitting to bleed pressure then remove test tee fitting" after the transmitter is removed from service. This step is performed to relieve RCS system pressure before connecting calibration equipment. Contrary to the instruction of this step, the IMs relieved system pressure from the sense line via the high side drain valve which is located at the lowest point in the sense line. The drain line routes to a closed drain system which made it impossible to determine how much fill fluid was lost when the drain valve was opened momentarily. During the investigation, it was theorized that opening of the drain valve allowed a sufficient amount of RCS water to be drained out of the sense line such that on the last step of return to normal when the transmitter high side isolation valve was opened, the void in the high side drain line caused a pressure drop in the common sense line to 2-FT-68-71D. This pressure drop caused a momentary decrease in the 2-FT-68-71D output, subsequent actuation of bistable 2-FS-68-71D, and a reactor trip.

To substantiate this theory, Work Request (WR) B261158 was initiated to reenact the exact work sequence, including the procedure deviation, for valving out and valving in of 2-FT-68-71B. Recorder charts were connected to the loop test point on loop 2-F-68-71D to monitor the loop current signal during the reenactment.

The performance of the WR showed that upon return to normal of transmitter 2-FT-68-71B (i.e., opening of the high side isolation valve), the output of 2-FT-68-71D decreased below setpoint of 2-FS-68-71D. The setpoint of 2-FS-68-71D is 36.78mA and the output of 2-FT-68-71D decreased to approximately 34.OmA. This would have caused the 2-FS-68-71D bistable to actuate, simulating a low flow condition and would have completed the necessary reactor trip logic. This reenactment along with the prior successful performance of SI-246 steps for transmitters 2-FT-68-6A, -6B, -29A, and -71A proved the theory as discussed above.

CAUSE OF EVENT

This reactor trip was a result of a RCS loop 4 low flow signal (two out of three channels tripped) coincident with permissive P-8 (above 35 percent power). The RCS loop 4 low flow reactor trip signal was caused by channel II bistable 2-FS-68-71B being placed in the tripped condition according to SI-246 and the subsequent inadvertent actuation of channel III bistable 2-FS-68-71D (completed the two out of three logic). The cause of the 2-FS-68-71D inadvertent actuation is attributed to a SI-246 procedure deviation while performing calibration on 2-FT-68-71B.

ANALYSIS OF EVENT

This event is being reported in accordance with 10 CFR 50.73, paragraph a.2.iv, as an event which resulted in the automatic actuation of the reactor protection system.

All safety-related equipment actuated as designed to mitigate the reactor trip event. The reactor trip breakers opened and all control rods dropped to bottom position as designed. Feedwater isolation occurred from the reactor trip coincident with RCS low average temperature (low Tavg) of 554 degrees F and letdown isolation occurred at 17 percent pressurizer level. All three auxiliary feedwater (AFW) pumps started as designed, and the turbine-driven AFW pump was shut down at approximately 0136 EDT and placed in automatic control. The auxiliary boiler was started to supply steam to the sealing steam system.

Operations personnel responded exceptionally well during the reactor trip and subsequent transients. The operators response demonstrated a thorough knowledge of plant systems and the ability to control plant transients in a safe manner.

The event investigation team responded immediately after the reactor trip to determine cause and any adverse consequences/equipment malfunctions of the trip. The event investigation program is controlled by plant procedures SQA-186, "Root Cause Assessment for Adverse Actions/Conditions," and Administrative Instruction (AI)-18, Appendix A, Part B, "Reactor Trip Report," to ensure an orderly and organized investigation.

CORRECTIVE ACTIONS

An immediate investigation was initiated to determine the cause of the reactor trip and to determine if any corrective actions would be necessary before returning the unit to operation.

The remaining RCS flow transmitters which required recalibration as a result of the SI-155 data were recalibrated before 35 percent power (permissive P-8). This event was reviewed with Operations and Instrument Maintenance personnel to ensure lessons learned from this event were identified and to reemphasize the necessity of procedural compliance.

Additionally, the Plant Operations Review Staff (PORS) will review previous reactor trips at Sequoyah for a similar occurrence of a common interaction between equipment that has caused a reactor trip. This review will provide assurance that an adverse trend does not exist on the number of reactor trips from common equipment interactions and will help prevent recurrence of this type event. This review will be complete by August 23, 1988.

ADDITIONAL INFORMATION

This was the second reactor trip occurrence since unit 2 has star
ed up after
the extended shutdown of approximately 2 1/2 years (8/85 - 5/88). The first reactor trip detailed in LER SQRO-50-328/88023 was caused by an equipment malfunction.

COMMITMENTS

PORS will review previous reactor trips for a similar occurrenc of common equipment interactions causing reactor trips by August 23, 1988.

ATTACHMENT # 1 TO ANO # 8806290030 PAGE: 1 of 1

TENNESSEE VALLEY AUTHORITY
Sequoyah Nuclear Plant
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June 17, 1988

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNIT 2 - DOCKET NO.
50-328 - FACILITY OPERATING LICENSE DPR-79 - REPORTABLE OCCURRENCE REPORT
SQRO-50-328/88024

The enclosed licensee event report provides details concerning a unit 2 reactor trip on a low reactor coolant flow signal caused by a procedure noncompliance. This event is reported in accordance with 10 CFR 50.73, paragraph a.2.iv.

Very truly yours,
TENNESSEE VALLEY AUTHORITY
/s/ S. J. Smith
S. J. Smith
Plant Manager

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